NON-PUBLIC?: N
ACCESSION #: 8810310522
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000410

TITLE: Manual Reactor Scram Necessitated by Loss of Reactor Building Closed Loop Cooling Due to Personnel Error
EVENT DATE: 09/22/88 LER #: 88-051-00 REPORT DATE: 10/21/88

OPERATING MODE: 1 POWER LEVEL: 064

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: R.G. Smith
Operations Superintendent TELEPHONE: 315-349-2388

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE TO NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On September 22, 1988 at 1339 hours, with the reactor at approximately 64% of rated thermal power and the mode switch in "RUN", Nine Mile Point Unit 2 (NMP2) experienced an actuation of an Engineered Safety Feature (ESF). Specifically, the ESF consisted of a manual reactor scram.

As a result of a post maintenance valve stroke on service water valve 2SWP*MOV18B, the Service Water System (SWP) was cross connected to the Reactor Building Closed Loop Cooling System (CCP). This cross connection caused loss of CCP inventory to the service water discharge header, which eventually resulted in the trip of CCP booster pumps on low suction pressure. Upon loss of CCP, the Station Shift Supervisor ordered a manual scram in accordance with N2-OP-13, "Reactor Building Closed Loop Cooling".

The cause of this event was personnel error due to inattention to detail. Operations personnel failed to assess the impact of opening 2SWP*MOV18B with the existing system configuration.

Corrective actions are as follows: 1) CCP was returned to normal and hold out tags were placed on the CCP and SWP valve controls; 2) An operator aid has been added to the control room panels warning of the impact of opening the service water crosstie valves; 3) A lessons learned has been issued stressing the importance of assessing plant impact.

End of Abstract

I. DESCRIPTION OF EVENT

On September 22, 1988 at 1339 hours, with the reactor at approximately 64% of rated thermal power and the mode switch in "RUN", Nine Mile Point Unit 2 (NMP2) experienced an actuation of an Engineered Safety Feature (ESF). Specifically, the ESF consisted of a manual reactor scram.

The NMP2 Spent Fuel Pool Cooling (SFC) heat exchangers normal cooling water supply is provided by the Reactor Building Closed Loop Cooling System (CCP) Attachment 1. The SFC heat exchanger CCP inlet valve is normally open
nd the
outlet valve normally closed. The service water system provides a backup cooling water supply and is normally isolated at the SFC heat exchanger inlet and outlet.

Electrical Maintenance personnel, working under Work Request (WR 110162) adjusted the torque switch of Service Water,(SWP) valve 2SWP*MOV18B. 2SWP*MOV18B is the SWP discharge valve from spent fuel heat exchanger 2SFC*ElB. After the adjustment was completed, Maintenance personnel requested the control room to stroke test the valve as directed by procedure N2-EMP-GEN-510, "Limitorque Disassembly and Assembly of Type SMB and SB Series Operators". The valve was then stroked as directed.

The result of stroking the SFC heat exchanger SWP outlet valve was to cross connect the reactor building closed loop cooling system with SWP through the SFC heat exchanger. Opening the SWP outlet valve provided a flow path for CCP through the open CCP inlet valve, to the SFC heat exchanger, out the SWP outlet valve and ultimately to the SWP discharge bay. This caused the CCP booster pumps to eventually trip on low suction pressure.

Upon total loss of CCP, the Station Shift Supervisor ordered a manual scram in accordance with operating procedure N2-OP-13, "Reactor Building Closed Loop Cooling" Section H.1, "Loss of Reactor Building Closed Loop Cooling".

There were no other out of service/inoperable components which contributed to this event.

II. CAUSE OF EVENT

The root cause of this event is personnel error due to inattention to detail. A Licensed Operator opened SFC heat exchanger SWP outlet valve 2SWP*MOV18B without assessing the plant impact.

The following were contributing causes to the event:

1. The protective tagging for the required maintenance was not adequate to cover the maintenance test.

2. The Work Request (WR 110162) was permitted to come to the control room without an adequate "pre-review ". The "pre-review" or "initial plant impact assessment" on work requests for all Maintenance Departments is performed by the applicable Departments planning group. The planning groups consist of maintenance personnel. Maintenance personnel are not extensively trained in plant systems and are therefore, in some cases, not qualified to make an impact assessment. An adequate impact statement becomes especially important when using generic procedures, such as N2-EMP-GEN-510, which do not have specific impact statements. Scheduled work is reviewed by Operations management, but not to a level of detail which determine adequacy of protective tagging.

3. The SFC heat system had been out of service for over a year, and an evolution involving cooling water to the heat exchanger was considered non-impacting.

III. ANALYSIS OF EVENT

A manual reactor scram is a conservative event and poses no adverse safety consequences at any reactor power. This event did not adversely affect any safety system nor the operators' ability to achieve safe shutdown.

The CCP system is designed to remove heat from various auxiliary equipment housed in the reactor building and turbine building during normal plant operation. During emergency or faulted plant conditions portions of the system provide a category I pressure boundary for backup cooling from the service water system to cool the SFC heat exchangers and residual heat removal pump seal coolers. However, the CCP system is not required to operate during an emergency or faulted plant condition. Therefore, a loss of CCP posed no adverse safety effects.

During normal plant operation the CCP system provides an intermediate barrier between systems containing radioactive products and the Service Water System (SWP). The SWP transfers CCP heat load to the ultimate heat sink. The system configuration which led to this event allowed CCP water to be directed to the SWP discharge bay. Therefore, the potential for a radioactive release did exist.

The CCP system was out of service for approximately 26 minutes.

IV. CORRECTIVE ACTIONS

1. Initial actions were to return the reactor building closed loop cooling system to normal and place holdout tags on effected CCP and SWP valve controls to warn of further operation.

2. Samples of CCP water were taken and checked for radiation. No abnormal radiation levels were found.

3. An operator aid has been added to the control room panels warning of the impact of opening CCP to service water crosstie valves.

4. A lessons learned transmittal has been issued to the Electrical Maintenance, Mechanical Maintenance, Instrument & Control, and Operations Departments. The transmittal stresses the importance of assessing the potential plant impact prior to allowing a job to start.

5. Niagara Mohawk management has evaluated the need to add systems qualified personnel to the NMP2 work control group. A work control group consisting of Maintenance and systems qualified personnel will be formed to provide a more detailed pre-review of work entering the control room.

6. A review of plant systems by the Operations Department revealed several instances where system crossties could cause plant problems. All of these potential problems were dispositioned prior to plant restart. This included locking shut the SFC heat exchanger inlet and outlet SWP valves and changing the affected procedure valve lineup.

7. Plant Engineering Support will perform a formal review of all plant systems for potential crosstie problems. The review will include:

*Identification of all system crossties

*Identification of protection against potential problems (i.e., interlocks, check valves, etc.)

*Potential plant impact of any crossties without protection

*Possible or suggested resolution of any identified problems

This review will be documented and a report generated providing a summary of results. This report will be provided to Operations for their use.

V. ADDITIONAL INFORMATION

Identification of Components Referred to in this LER

IEEE 803 IEEE 805
Component EIIS Funct System ID

Spent Fuel Pool Cooling N/A DA
Reactor Building Closed Loop Cooling N/A CC
Service Water N/A BI
Motor Operated Valve 20 BI
Pump P CC
Heat Exchangers HX DA

Failed Components - None

Previous Similar Events - There are previous events where the cause was at least partially related to lack of plant impact assessment. These events are detailed in LER 87-17, 87-26, 87-64, 88-06, and 88-17. The corrective actions for these LERs, except 88-17, addressed immediate and problem specific causes and did not focus on administrative procedure AP-5.2, "Unit 2 Procedure for Repair". AP-5.2 controls the work control process. LER 88-17 corrective action requires AP-5.2 and AP-3.3.2, "Control of Equipment Temporary Modifications" to be revised to clearly address plant impact requirements. These revisions, however, had not been implemented previous to the event described in this LER.

ATTACHMENT 1 TO 8810310522 PAGE 1 OF 1

NOT KEYABLE - DRAWING

Reactor Building
Closed Loop Cooling

ATTACHMENT 2 TO 8810310522 PAGE 1 OF 1

NIAGARA
MOHAWK NMP 41608

NINE MILE POINT-UNIT 2/P.O. BOX 63, LYCOMING, NY 13093/TELEPHONE
(315) 343-2110

October 21, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 88-51

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 88-51 Is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv), "Any operation or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR50.72 (b)(2)(ii) report was made at 1516 hours on September 22, 1988.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

J.L. Willis
General Superintendent
Nuclear Generation

JLW/JMT/mjd

Attachments

cc: Regional Administrator Region 1
Sr. Resident Inspector: W. A. Cook

*** END OF DOCUMENT ***
